ANL-7117

ANL-7117

Argonne National Laboratory

INSPECTION, EVALUATION, AND OPERATION OF THE EBWR REACTOR VESSEL

by

N. Balai, C. R. Sutton,
E. A. Wimunc, and R. F. Jones

Printed in USA. Price $3.00. Available from the Clearinghouse for Federal Scientific and Technical Information, National Bureau of Standards, U. S. Department of Commerce, Springfield, Virginia

ANL-7117
Reactor Technology
(TID-4500, 46th Ed.)
AEC Research and
Development Report

ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue
Argonne, Illinois 60440

INSPECTION, EVALUATION, AND OPERATION OF THE EBWR REACTOR VESSEL

by

N. Balai, C. R. Sutton, E. A. Wimunc
Reactor Engineering Division

and

R. F. Jones
Reactor Operations Division

November 1965

Operated by The University of Chicago
under
Contract W-31-109-eng-38
with the
U. S. Atomic Energy Commission

TABLE OF CONTENTS

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF FIGURES

LIST OF TABLES

INSPECTION, EVALUATION, AND OPERATION OF THE EBWR REACTOR VESSEL

by

N. Balai, C. R. Sutton,
E. A. Wimunc, and R. F. Jones

ABSTRACT

A program was initiated to operate the Experimental Boiling Water Reactor (EBWR) with a plutonium fuel loading as part of the Commission's Plutonium Recycle Program. The reactor facility had been completed in 1956 and was in operation until the 100-MWt experimental program was finished. Preparations for operating the EBWR in this new program included the examination of the reactor vessel. The interior upper half of the vessel was accessible for inspection (the lower half being inaccessible) where the exposed surface of the cladding was subjected to visual and dye-penetrant examination. Minor cracks and surface decorations were found in the stainless-steel Type 304 cladding; in many instances, gas tests revealed them to be through cracks.

The through cracks were ground out to the pressure-vessel base metal (SA-212-B) with dye-penetrant inspections interspersed during the process. The grinding and inspection were extended to, and slightly into, the SA-212-B base metal. No evidence was found of crack propagation into the base metal; nor was there evidence of local pitting or gross generation of iron oxide (Fe_2O_3) at the clad base metal interface.

As a result of these preliminary findings, a more extensive investigation was initiated to assure that cracks did not propagate into, or develop in, the SA-212-B carbon-steel pressure vessel. This included the removal of boat samples from the reactor vessel for metallographic and chemical analysis, and the examination of similar and related materials. This report describes the findings of this investigation, their significance, and the considerations relating to the future operation of the reactor.

I. INTRODUCTION

A. History of EBWR

The Experimental Boiling Water Reactor (EBWR) facility was constructed to demonstrate the feasibility of operating an integrated power plant utilizing a boiling-water reactor as the heat source. The reactor was designed to produce 20 MW of heat in the form of 600-psig saturated steam, which was fed directly to a turbogenerator, producing 5 MW of electricity. Full-power operation at its design condition of 20 MWt was realized in December 1956.

Subsequent experiments at power levels up to 40 MWt indicated that stable operation at powers as high as 66 MWt was possible with the initial 4-ft-diam core. This was affirmed in 1958 by a short-term experimental operation of the plant at 61.7 MWt. Further power increase at this time was precluded by the feedwater pumps, which were operating at maximum capacity.

Analyses and projection of the experimental data to a 5-ft-diam core indicated that, with some modification of the core structure and pressure-vessel internals and with additional heat-removal equipment, EBWR could operate at or near 100 MWt.

In 1960, modifications were made to the plant to increase its power-handling capability to 100 MWt. This included installation of several new nozzles in the reactor vessel, including a 6-in. feedwater line and a 10-in. steam line. In 1962, the plant was operated at 100 MWt, and upon completion of the experimental program, the plant was shut down and the boiling-water program at the EBWR was terminated.

In anticipation of release of the EBWR from the Water Cooled Reactors Program, the Division of Reactor Development early in 1962 requested that Argonne participate in the Plutonium Recycle Program with particular emphasis on the portion of the work that would be appropriate using the EBWR facility. The preparations for operating EBWR in this program included the examination of the reactor vessel.

Initially, visual and dye-penetrant inspections of the accessible portions of the interior of the vessel revealed numerous cracks in the resistance-welded cladding of the cylindrical part of the vessel. The removal of the steam-collector duct and the shock shield permitted inspection of approximately half (upper course) of the reactor vessel. All cracking was found to be in the resistance-welded cladding. The upper ring forging, vessel head, and nozzles, which were clad by weld-overlay techniques, were free of cracks.

Background information on the reactor pressure vessel relative to its fabrication, cladding process, and past operating history follows.

B. History of Reactor Pressure Vessel

1. Reactor Vessel Fabrication

The EBWR pressure vessel was designed and fabricated in 1955-1956 in accordance with Section I, "Power Boilers," of the ASME Boiler and Pressure Vessel Code (1952 Revision). Details of design and construction are summarized in ANL-5607,[1] and the more pertinent details of forming, welding, and stress relief are described in previous reports.[2,3,4] The vessel was fabricated from SA-212-B plate, clad with AISI Type 304 stainless steel.

2. Cladding Process and Integrity

The vessel plates for the shell and lower head were clad by intermittent spot-resistance welding of 0.109-in.-thick, stainless-steel Type 304 sheet to SA-212-B plate. The plate sections were covered with panels of cladding (approximately 32 in. wide), which were later manually butt-welded to yield a continuous clad surface over the SA-212-B vessel steel. The cladding was resistance-welded to the base plate while both were submerged in water. A flow of water was maintained to the tank as makeup for losses due to spillage or evaporation. The square, lower head plate was clad in a similar manner before being cut into a circular blank for the head-forming operation.

Integrity of the cladding on the finished vessel was established by leak testing. The test was performed by introducing nitrogen at 900 psi pressure between the cladding and the plate and by utilizing liquid soap on the exposed surface of the cladding. Some leakage (cracks) was detected; the cracks were repaired by welding prior to shipping the vessel.

The upper-ring forging and vessel head were clad by the submerged-arc process, which deposited an apparently crack-free layer of cladding.

3. Operating Conditions and Duration

Initial operation of the reactor vessel at 600-psig saturated (489°F) condition occurred in 1956. Operations under this condition were intermittent up to 1963 when power levels of 100 MWt were attained. However, the major portion of the operations was at 20 MWt.

Several new nozzles were installed in the reactor vessel during the 100-MWt conversion in 1959-1960. When the vessel cladding was inspected (by using dye-penetrant) during this modification, no defects in the cladding were found.

C. Nondestructive Inspection

A nondestructive inspection program of components in the reactor primary system was undertaken in December 1964 and included the reactor pressure vessel. Visual examinations revealed numerous surface decorations on the resistance-welded cladding.

When these cracks were found, the inside of the reactor vessel was subjected to a detailed inspection. To gain accessibility to the entire upper half of the reactor-vessel cladding, the shock shield and the steam-collecting duct were removed. Through cracking of the cladding was definitely established by a gas-leak test.

D. Summary of Initial Findings

The results of nondestructive inspection on the cladding, visual, and dye-penetrant, are summarized below. Also included are the results of examinations of clad specimens which are similar or identical to the vessel.

As a consequence of these initial observations, a thorough investigation was undertaken which utilized destructive examinations of samples to determine whether these cracks propagated into, or were present in, the SA-212-B steel vessel wall and to establish, if possible, the mechanism of cracking.

1. Reactor Vessel Cladding

The upper course of the vessel is covered by eight vertical panels of cladding joined by longitudinal welds. Each panel is approximately 32 in. wide by 110 in. long. Five of the eight panels contained surface cracks. Over 75% of the cracks were in the vicinity of the steam zone and were clustered between the weld nuggets adjacent to the arc-deposited welds joining the panel sections (see Fig. 1).

None of the submerged arc-deposited cladding of the conical ring forging and vessel head was cracked.

2. Four-inch-thick Lower-head-plate Remnant

The cusp-shaped corner pieces (remnants) that were cut from the lower head plate before it was formed into an ellipsoidal head, were inspected. These pieces were salvaged from the initial vessel fabrication (1955). Dye-penetrant tests of the cladding revealed minor surface cracks similar to those found in the reactor vessel.

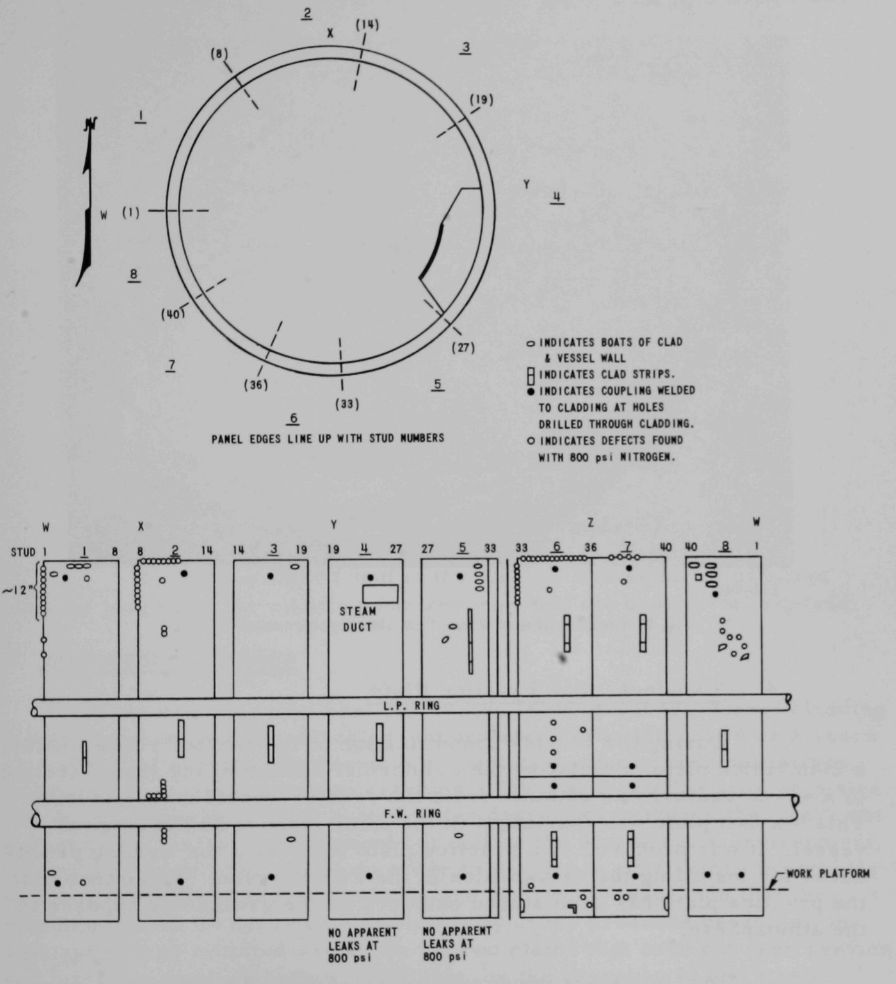


Fig. 1. Orientation of Reactor-vessel Cladding Panels Showing Location of Boat and Strip Specimens

3. Crescent-shaped Vessel Plug

A crescent-shaped section of the reactor vessel wall, retained from vessel modifications of 1960 in preparing the facility for 100-MWt operation, was inspected during this period (1965). Dye-penetrant tests revealed that the surface of the cladding was free of cracks. The cladding on this plug was part of vessel panel No. 5 which was also free of surface cracks (see Fig. 2).

144-746

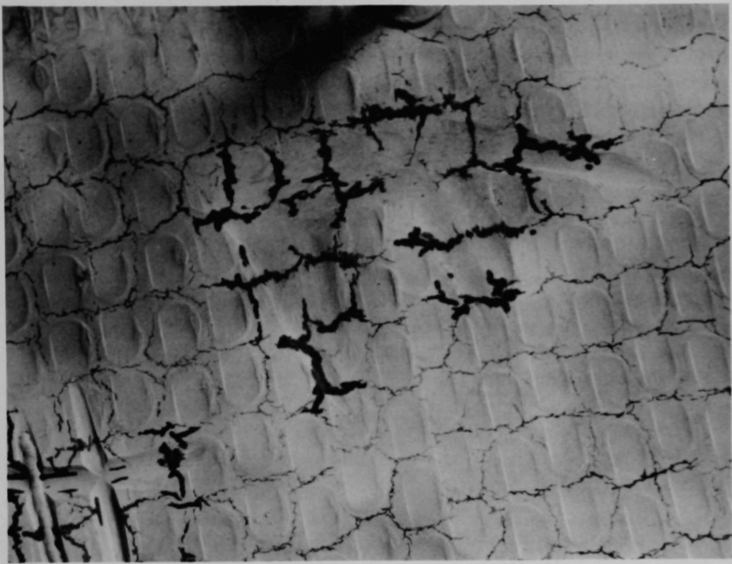
Fig. 2. Crescent-shaped Vessel Plug after Dye-penetrant Testing

4. Two-inch-thick Practice Plate

During the 100-MWt modification of the reactor vessel in 1960, a 2-in.-thick plate, simulating the cylindrical section of the vessel (rolled to a 42-in. radius) was obtained from the reactor-vessel manufacturer. This 4 x 8-ft plate was fabricated in the same manner as the original vessel. It was procured as a practice plate to develop the welding procedures for installing the new nozzles in the EBWR vessel. Since that time, the practice plate has been stored outdoors on the ground and exposed to the atmosphere.

Dye-penetrant inspection revealed the clad surface on this plate to be more extensively cracked than the reactor vessel cladding, as shown in Fig. 3. In addition, attempts to perform practice welds on the cladding resulted in new crack formation and crack extension in the cladding.

Items 2 and 4 described above have never seen reactor operating service or been exposed to a reactor environment.



144-732

Fig. 3. Two-inch-thick Practice Plate after Dye-penetrant Testing

No cracks were found in the SA-212-B base metal of either plate or plug when the cladding was removed and the base metal exposed.

E. Statement of Problem

When cracking was first discovered in the pressure-vessel cladding it was assumed that this cracking had developed over the past 5 or 6 years as a result of service conditions, e.g., radiation effects, thermal fatigue, and possible stress-corrosion cracking. The discovery that similar plates had failed, apparently to an even greater degree, even though they were not exposed to a reactor environment but only to the atmospheric environment, was cause for re-evaluation. In order to develop a satisfactory hypothesis to account for the observed cracking and to provide assurance that the cladding cracks do not impair the integrity of the pressure vessel, an investigation was initiated--first, on the two plates that have not seen service, and later, as samples became available, on the pressure vessel itself. This involved inspections of the destructive type. An extended investigation was carried out primarily to determine whether the cladding cracks propagated into the SA-212-B pressure-vessel base metal. Complete details of nondestructive and destructive test results are given in Section II.

II. INVESTIGATION AND RESULTS

A. Nondestructive Examinations

Two defects (holes) were spotted on the cladding of the vessel when a working platform was lowered into the reactor vessel to enable inspection of the conical ring forging. A streak of rust, deposited on the cladding surface, was observed, which upon closer inspection revealed two through-holes in the cladding. This observation prompted a thorough inspection of the cladding in the upper half of the vessel available for inspection.

Nondestructive examination of cladding in the reactor vessel included visual observation and dye-penetrant tests.

1. Conical Ring Forging

Ultrasonic longitudinal and shear wave examinations were used to establish the integrity of the highly stressed portion of the ring forging. The search patterns, on the vessel exterior, included:

a. Scans of the inter-bolt hole ligaments in the longitudinal direction.

b. Scans of the inter-bolt hole ligaments in the radial direction.

c. Circumferential scans of planes inscribing and circumscribing bolt holes in the longitudinal direction.

d. Circumferential scan of plane immediately below the bolt hole bottoms in the radial direction.

There was no evidence of cracking or material faults (as lamination).

The arc-deposited cladding was examined with dye-penetrant for possible cracking; none was detected.

2. Closure Bolting

All 44 bolt studs, threaded into the conical-ring forging, were examined by the ultrasonic-longitudinal-wave and magnetic-particle techniques. All closure bolts were found to be free of longitudinal, radial, and thread-root cracks.

3. Vessel Cover Plate

The cover plate was examined by ultrasonic longitudinal waves in the radial and thickness directions for inter-ligament and concentric laminar cracking. In addition, every eighth bolt hole was magnetically examined for incipient edge cracking. The closure plate was found to be sound and free of cracks.

4. Cylindrical Section

The upper course (one-half) of the cylindrical section was examined, and indications of gross cracking of the spot-welded cladding were noted. The lower course and the lower head were inaccessible.

It was not clear from the visual and dye-penetrant inspections whether crack indications were simply surface indications or actual cladding cracks. To explore this condition further, destructive testing was inaugurated.

B. Destructive Examinations

1. In-situ Grinding Explorations of Vessel Cladding

As a preliminary to the destructive examinations, a pressure test similar to that conducted upon completion of the vessel fabrication was made. Nitrogen at 800 psig was introduced between the Type 304 stainless-steel cladding panels and the SA-212-B pressure shell. A soap-bubble-type test was conducted on the exposed face with an aqueous foaming detergent. The pressure test was most conclusive since it also revealed that the cracks breached completely through the (resistance-welded) cladding in many areas. These cracks existed in five of the eight panels (see Fig. 1) comprising the internal cladding in the upper cylindrical half of the vessel. Initially no through cracking was found in the three "good" panels. Subsequent dye-penetrant testing revealed only a small amount of very superficial cracking in two of these panels. The third panel had no indication of cracking.

The arc-deposited weld-overlay cladding, which was applied to the upper-ring forging, the underside of the vessel-head cover, and areas around vessel nozzles, was also examined with the aid of a dye-penetrant. No defects were found.

All the cladding panels were observed throughout the pressure tests to guard against peeling of the cladding from the base metal. There was no evidence of cladding separation or other signs of failure at pressures to, and including, 800 psi.

Almost all of the cracks followed paths between weld nuggets, as shown in Fig. 4 (thumb print). A few cracks, however, either ran across a "thumb print" or terminated in one.

II. INVESTIGATION AND RESULTS

A. Nondestructive Examinations

Two defects (holes) were spotted on the cladding of the vessel when a working platform was lowered into the reactor vessel to enable inspection of the conical ring forging. A streak of rust, deposited on the cladding surface, was observed, which upon closer inspection revealed two through-holes in the cladding. This observation prompted a thorough inspection of the cladding in the upper half of the vessel available for inspection.

Nondestructive examination of cladding in the reactor vessel included visual observation and dye-penetrant tests.

1. Conical Ring Forging

Ultrasonic longitudinal and shear wave examinations were used to establish the integrity of the highly stressed portion of the ring forging. The search patterns, on the vessel exterior, included:

a. Scans of the inter-bolt hole ligaments in the longitudinal direction.

b. Scans of the inter-bolt hole ligaments in the radial direction.

c. Circumferential scans of planes inscribing and circumscribing bolt holes in the longitudinal direction.

d. Circumferential scan of plane immediately below the bolt hole bottoms in the radial direction.

There was no evidence of cracking or material faults (as lamination).

The arc-deposited cladding was examined with dye-penetrant for possible cracking; none was detected.

2. Closure Bolting

All 44 bolt studs, threaded into the conical-ring forging, were examined by the ultrasonic-longitudinal-wave and magnetic-particle techniques. All closure bolts were found to be free of longitudinal, radial, and thread-root cracks.

3. Vessel Cover Plate

The cover plate was examined by ultrasonic longitudinal waves in the radial and thickness directions for inter-ligament and concentric laminar cracking. In addition, every eighth bolt hole was magnetically examined for incipient edge cracking. The closure plate was found to be sound and free of cracks.

4. Cylindrical Section

The upper course (one-half) of the cylindrical section was examined, and indications of gross cracking of the spot-welded cladding were noted. The lower course and the lower head were inaccessible.

It was not clear from the visual and dye-penetrant inspections whether crack indications were simply surface indications or actual cladding cracks. To explore this condition further, destructive testing was inaugurated.

B. Destructive Examinations

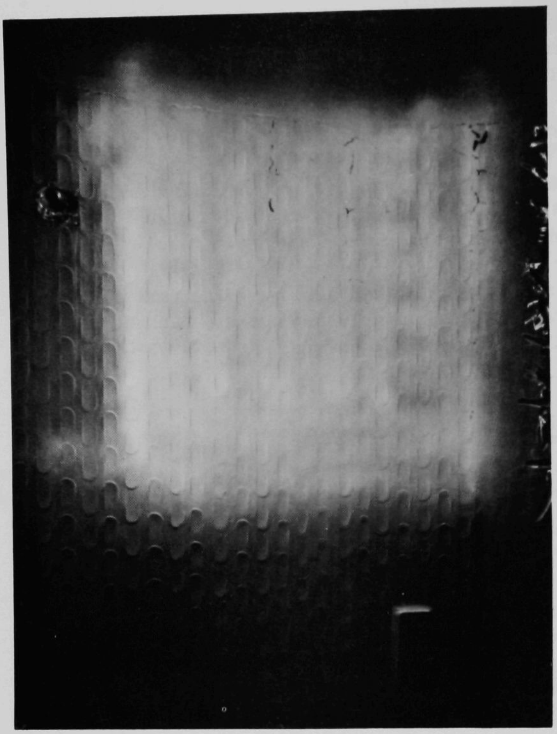
1. In-situ Grinding Explorations of Vessel Cladding

As a preliminary to the destructive examinations, a pressure test similar to that conducted upon completion of the vessel fabrication was made. Nitrogen at 800 psig was introduced between the Type 304 stainless-steel cladding panels and the SA-212-B pressure shell. A soap-bubble-type test was conducted on the exposed face with an aqueous foaming detergent. The pressure test was most conclusive since it also revealed that the cracks breached completely through the (resistance-welded) cladding in many areas. These cracks existed in five of the eight panels (see Fig. 1) comprising the internal cladding in the upper cylindrical half of the vessel. Initially no through cracking was found in the three "good" panels. Subsequent dye-penetrant testing revealed only a small amount of very superficial cracking in two of these panels. The third panel had no indication of cracking.

The arc-deposited weld-overlay cladding, which was applied to the upper-ring forging, the underside of the vessel-head cover, and areas around vessel nozzles, was also examined with the aid of a dye-penetrant. No defects were found.

All the cladding panels were observed throughout the pressure tests to guard against peeling of the cladding from the base metal. There was no evidence of cladding separation or other signs of failure at pressures to, and including, 800 psi.

Almost all of the cracks followed paths between weld nuggets, as shown in Fig. 4 (thumb print). A few cracks, however, either ran across a "thumb print" or terminated in one.



144-738

Fig. 4

Panel Showing "Thumb Prints" Characteristic of Method of Cladding, Dye-penetrant Indications of Cracking and Half-coupling Used in Pressure Testing

A majority of the through cracks were ground out to the base metal with dye-penetrant inspections interspersed in the process to examine the character of the cracks through the cladding. The cracks appeared to increase in size and in number as the cladding was ground out, suggesting that the cracks may have originated at the interface between the cladding and the base metal. This was generally true for cracks that extended all the way through the cladding, but it cannot be definitely stated for all such cracks since it was found that many of the cladding cracks started at a point remote from the interface, such as at the weld nugget. It should be pointed out, however, that this characteristic of the cracks could also result from a changing stress pattern in the cladding as material was removed. Approximately 40 linear feet of cracks were ground out and examined in the above manner. The surface was carefully cleaned, and dye-penetrant testing was continued into the base metal (approximately 0.010 to 0.015 in. deep). No evidence of cracking of the base metal was found. There was also no evidence of parting in the resistance welds bonding the cladding to the vessel wall.

Approximately 30 sq in. of the SA-212-B steel behind the cladding became available for examination when strips of cladding were removed for chemical analyses. Other areas of the shell wall were uncovered when the exploratory grinding operations removed fragments of cladding bounded by crack interfaces. The appearance of the panel having the most amount of metal ground away is shown in Fig. 5. In all exposed areas, the carbon-steel wall was found to be covered with a thin film of black oxide (probably Fe_3O_4). No pitting attack or red oxide (Fe_2O_3) was found. The interface side of the cladding, nominally in contact with the SA-212-B, was also covered with a thin, adherent, black oxide film and was free of reddish-brown iron rust.

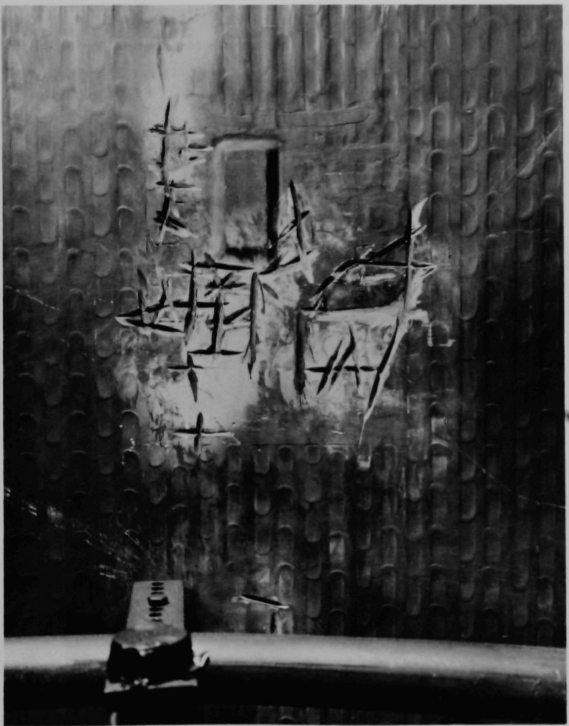


Fig. 5

Appearance of the Panel Having Greatest Amount of Cladding Ground Away

144-736

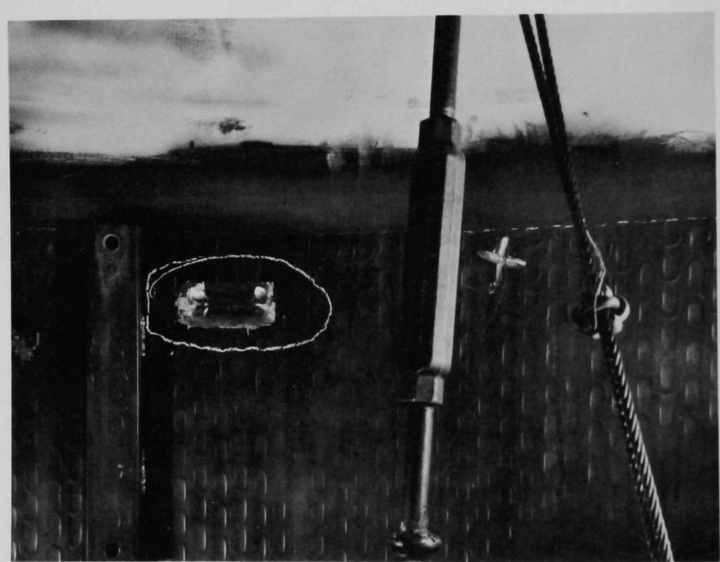
2. Vessel Samples

The types of cladding samples removed from the wall of the reactor pressure vessel for examination were (a) boat samples and (b) strip samples.

a. Boat Samples

Alundum grinding wheels were used to cut through the cladding. This was followed by two 45° angle cuts into the SA-212-B steel with a circular slitting saw ($1\frac{1}{2}$ in. diam by 0.025 in. thick). The object was to obtain some of the base metal along with the cladding for metallographic examinations in determining whether any cracks in the base metal were present. Infringement on the SA-212-B (base metal) wall did not exceed 1/8 in. in depth.

Boat samples, approximately 1/2 in. wide by $2\frac{1}{2}$ in. long (see Fig. 6) were cut from the upper portions of panels 1, 3, 5, and 8 to obtain representative samples of different degrees of cracking in the cladding. Panel 1 contained the most cracks, and panel 8 the least. Metallographic examinations were made on these boat samples. The results of these, as well as samples obtained from other supporting materials, are reported later.



144-804

Fig. 6. Typical Panel after Removal of Boat Sample

b. Strip Samples

Strip samples of cladding approximately 1/2 in. wide by 8 in. long were removed from the vessel wall for chemical analyses. These samples were removed from all panels. Sampling areas between adjacent rows of spot-welds were chosen so that cladding material did not contain any SA-212-B carbon steel.

The chemical analyses of these vessel-cladding samples are tabulated in Table I, along with the analyses of samples from the 2-in. practice plate, the 4-in. remnant plate, and the permissible ASME code values.

3. Other Materials Inspected

a. Four-inch-thick Lower-head-plate Remnant

This plate contained cracks in its cladding. Samples of the cladding were milled off for chemical analyses, and the results are tabulated in Table I.

b. Two-inch-thick Practice Plate

The history of this plate was presented earlier, along with findings on preliminary examinations, which revealed numerous surface cracks in the cladding. Portions of the stainless-steel cladding were removed for determining the chemical composition. Scrapings were taken from the interface surfaces to determine the chloride content. Results of these samplings are tabulated in Tables I and II, respectively.

c. Reactor-vessel Plug

The crescent-shaped section of the reactor-vessel wall, retained from vessel modifications of 1960, was inspected. Although no cracks were found in the cladding, chemical analyses were made on the interface oxide layer for chlorides. Results are given in Table II.

C. Supporting Examinations

1. Chemical Composition of Cladding Materials

Spectrochemical analyses of the cladding samples for the residual elements (Mo, W, Ti, Al, and Cu) showed none present in abnormal concentrations. No deleterious amounts of the heavy metals (Pb, As, Sn, Bi, and Sb) were found.

The chemical analyses revealed that the cladding met the Boiler Code Materials requirements except for sulfur. The slightly higher sulfur content of the vessel cladding contributes sulfide inclusions but is not believed to be otherwise injurious.

The results, together with the permissible range for Type 304 stainless steel, SA-240 plate material (1952 edition of Section II of the ASME Code) are presented in Table I.

TABLE I. Chemical Composition of Cladding-material Samples (%)

	From Reactor Vessel	From 2-in. Plate	From 4-in. Plate	Permissible SA-240 Plate Type 304
C	0.06-0.07*	0.050-0.046	0.060	0.08 max
Mn	1.48-1.51*	1.28	1.63	2.0 max
Si	0.46	0.38	0.53	1.0 max
Cr	17.95-18.29	17.5	18.8	18.0-20
Ni	9.98-10.8	8.7	9.8	8.0-12.0
S	0.035**			0.030 max

*Range of composition of samples from vessel wall.

**Average composition of nine samples from vessel wall.

2. Chloride Analysis

The cladding process at the fabricator's plant consisted of resistance-welding the stainless steel to the base plate while both were submerged in water. Softened water was used to remove the heat of welding. Since the softening process does not remove chlorides from the water, some chloride ions may have been trapped at the interface during the welding process.

Attempts were made to obtain samples of the thin oxide film on the SA-212-B vessel wall after the cladding was removed for chloride analysis. The extremely adherent, hard, dense film defied all removal efforts suitable for sampling. Chemical analysis of drillings from pressure-vessel boat samples were used instead and revealed the presence of chlorides as shown in Table II.

TABLE II. Summary of Chloride Analysis*

Source Material	Sample Location	Chloride Content, μg	Sodium Content	Sample Area, sq in.
Vessel boat samples	Panel 1--top	6 ± 5		~0.25
	--bottom	55 ± 25		
	Panel 3--top	60 ± 25		
	--bottom	425 ± 25		
	Panel 5--top	170 ± 25		
	--bottom	13 ± 5		
2-in.-thick practice plate	Approximately center of 20 x 24 in. section; one edge in contact with ground during 5 years of outdoor storage	5 ppm	1.6 ppm	~1.0
4-in.-thick lower-head-plate remnant		2.2 ppm	7 ppm	~1.0
Crescent-shaped vessel plug		0.6 ppm		~1.0

*By ANL Chemistry Division.

Samples of the interface deposit were also obtained from the 2-in.-thick practice plate, the lower-head-plate remnant, and the reactor-vessel plug. These samples were taken to determine whether a sufficient concentration of chlorides was present at the interface to cause stress-corrosion cracking at temperatures up to 120°F (August sun in Illinois). About one square inch of cladding in the unbonded region of each sample plate was milled away. The iron oxide at the interface was carefully collected and analyzed for chloride ion content. A 150-ml sample-plus-solvent showed 5 ppm of chlorides and 1.6 ppm of sodium. By making several assumptions (reasonable, but not necessarily valid) as to the thickness and density of the oxide layer, it has been calculated that the oxide itself contained about 3/4% chloride ion.

It is known from work by S. P. Rideout,[5] that some oxide films, notably Al_2O_3, have the ability to concentrate chlorides by large factors (in the order of 10 to 100). Concentration factors necessary to produce 3/4% Cl^- are, however, greater than would normally be expected.

Experiments were conducted at ANL to learn if Fe_2O_3 or Fe_3O_4 would adsorb chloride ion (Cl^-) preferentially from water containing sodium chloride.

Although there was a slight indication of adsorption of chloride by the Fe_2O_3, the amount was almost within the range of experimental error, so no definite conclusion can be drawn that selective adsorption has occurred. In any event, most of the cracks in the cladding of the 2-in. practice plate started at the outer surface of the stainless steel, apparently invalidating this mechanism in this case.

The analyses are believed to grossly over-estimate the actual contents of chlorides by virtue of accidental chloride contamination for reasons given in the following paragraphs.

a. <u>Vessel Boat Samples</u>

The boat samples were initially mounted in a cold-setting epoxy plastic for preliminary metallographic examinations. Later, these samples were mechanically recovered from the hardened plastic mount and dissolved in concentrated nitric acid for chloride determinations. Since chlorides are almost a universal contaminant, the resin and the polymerizing catalyst probably contained chloride contaminants. The wide variation in chloride contents (from 6 to 425 μg) are thus believed to reflect the variation in accidental contamination in the annuli of the boat sample which varied in thickness from zero in the bond region to 0.003 in. between neighboring spot-welds and between rows of spot-welds.

b. <u>Two-inch-thick Practice-plate Sample</u>

Cladding, of approximately one square inch in area, was removed from the plate by milling for the recovery of interface deposits. Unlike that of the reactor vessel, the deposit found was red-brown rust, which was recovered for analysis. The analysis shows a chloride-to-sodium ratio three times that for NaCl and indicates certain contamination. Since this plate was stored outdoors for several years, capillary forces had ample opportunities to concentrate natural and industrial contaminants from the ground and rainfall.

c. <u>Four-inch-thick Lower-head-plate Remnant</u>

Similar to the practice plate, the lower-head-plate remnant also had cladding removed by milling (approximately one square inch) for recovery of interface deposits. The deposits were found to be red-brown rust with a chloride-to-sodium ratio one-third that of the stoichiometric ratio for NaCl. This plate had also been stored outdoors for an even longer period of time than the 2-in.-thick practice plate.

d. Crescent-shaped Vessel Plug

The reactor vessel plug remaining from the 100-MW modifications to the vessel was also milled for recovery of interface deposits. In this case the deposits were black iron oxide with a much lower chloride content than the other samples. The plug had been exposed to the elements for only about a year, which substantiates the hypothesis that the 2-in. sample plate and the 4-in. lower-head-plate remnant were contaminated to some degree by outside storage.

The chloride content at the cladding interface was not established conclusively because the measured ratios of chloride to sodium are at wide variance with each other and with the stoichiometric ratios. The actual chloride concentration is believed to be substantially lower than indicated.

3. Metallography

Boat samples, approximately $2\frac{1}{2}$ in. long, were prepared for metallographic examination. These were taken from the reactor vessel shell (from panels 1, 3, 5, and 8--see Fig. 1) from the 4-in. lower-head-plate remnant, and from the 2-in. practice plate.

As-polished and etched samples were examined for details of cracking and metal structure in and near the weld nuggets. A 10% oxalic acid etch was utilized to reveal structural details in the SS Type 304 cladding; a 2% Nital etch brought out the details in the SA-212-B carbon steel.

Cracks in the AISI Type 304 stainless-steel cladding are intergranular except for a few hot tears in the interior of some of the weld nuggets.

Metallographic examination and Strauss Reagent testing shows that the stainless steel is severely sensitized; i.e., chromium carbide has precipitated at the grain boundaries to form an envelope. This phenomenon is brought about by subjecting austenitic stainless steel to temperatures in the range of 900-1500°F. The most critical temperature range is 1150-1200°F. The degree of precipitation is also a function of time. During stress relief of the vessel, the cladding was held at 1150-1200°F for at least 9 hr and was in the temperature range above 900°F for at least an additional 6 hr during heating and cooling.[4]

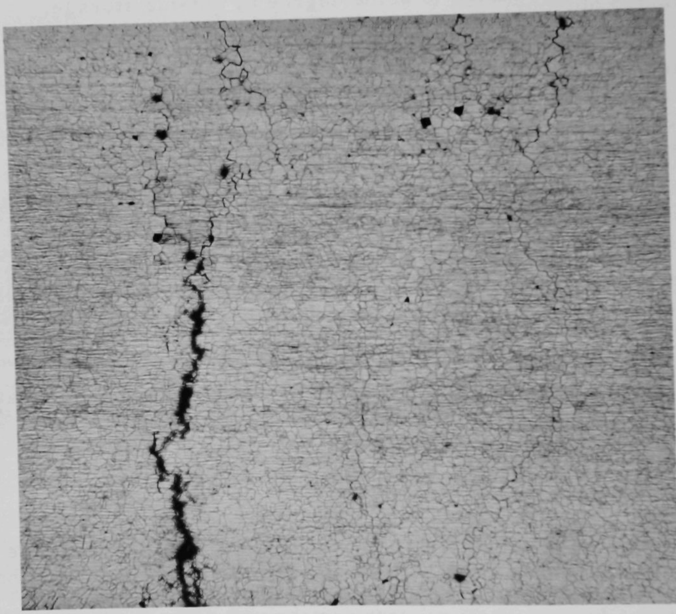
The metallographic data from SA-212-B plates, clad with SA-240 Type 304 stainless steel, are summarized and illustrated herewith.

a. Cracking Modes

(1) Vessel Cladding

Three types of internugget cracks were observed in the wrought cladding material.

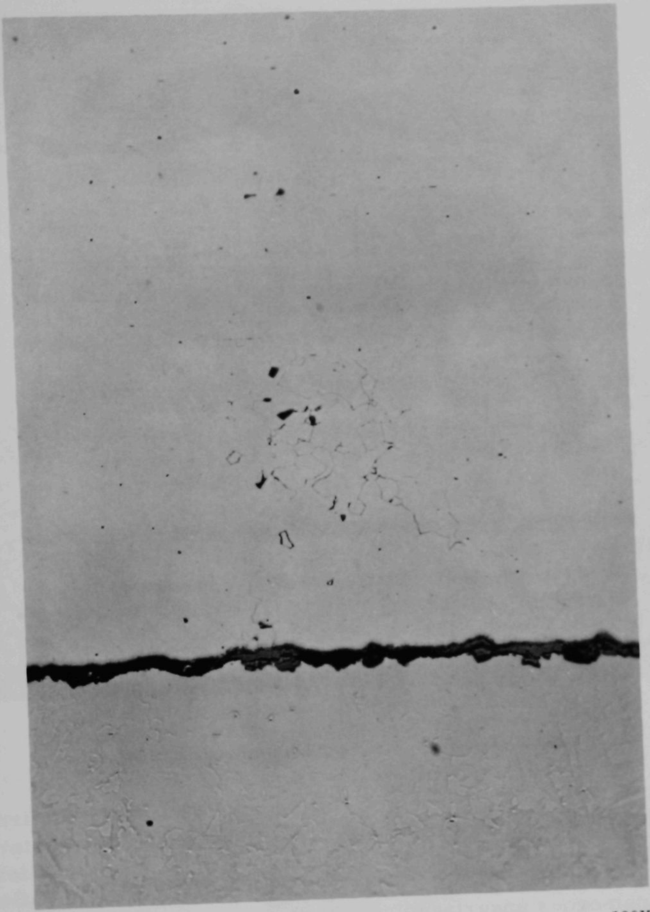
(a) Through cracks, which were wider at the interface, appearing to originate at the interface (Fig. 7).



42285 43X

Fig. 7. Through Cracks Apparently Originating at the SA-212-B/Type 304 Stainless-steel Interface

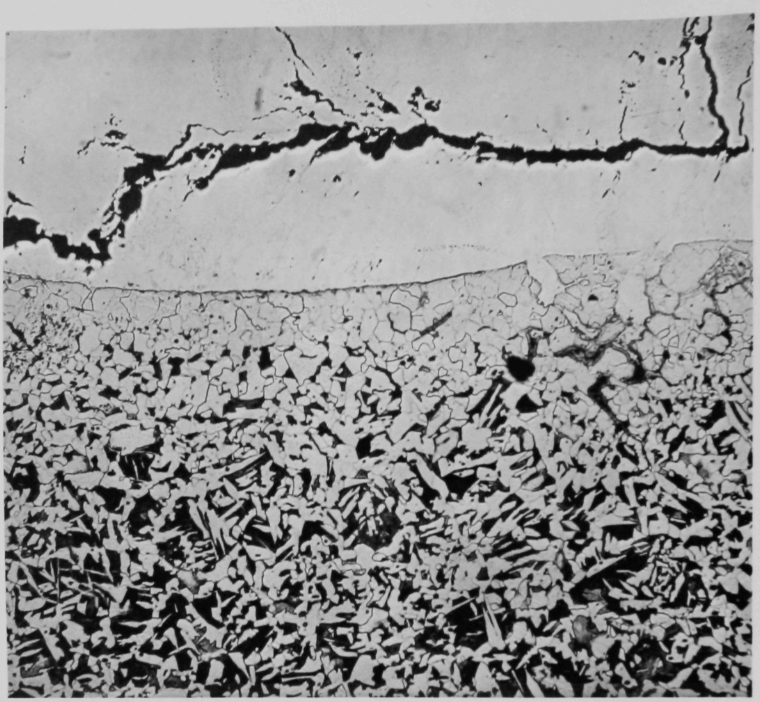
(b) Partially-developed and unsurfaced cracks, apparently unrelated to the presence of the through cracks (Fig. 8).



8636B 100X

Fig. 8. Unsurfaced Cracks Apparently Unrelated to Through Cracks

(c) Lamellar cracks, paralleling the cladding surfaces in apparently homogeneous metal, as well as in regions containing inclusions (Fig. 9).



42077 61X

Fig. 9. Lamellar Cracks, Paralleling the Cladding Surface

These metallographic studies largely confirmed the results of the destructive grinding explorations inside the reactor vessel. They revealed that the cladding from panel No. 5 which was believed to be "crack-free," as determined by surface dye-penetrant examinations, contained numerous unsurfaced cracks.

Examinations of the cladding nuggets revealed a region of great complexity in both the stainless-steel cladding and SA-212-B materials (see Figs. 9 and 10). In all cases, only the SS 304 cladding material in the nugget area had been melted by the passage of welding current. All the nuggets contained a dendritic kernel which was originally wrought cladding. Kernels were islands of cast material bounded by wrought stainless steel, Type 304 cladding, and the SA-212-B steel. The wrought cladding surrounding the cast kernel contained numerous unsurfaced cracks radiating from the kernel interface, as shown in Fig. 11. These cracks, which are usually normal to the surface of the kernel, are primarily due to grain-boundary liquation.

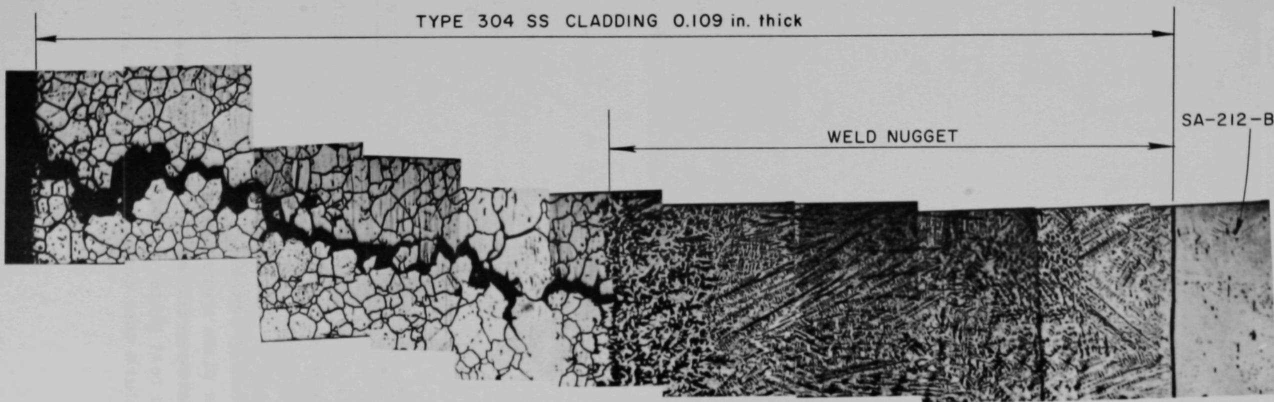


Fig. 10. Arrest of Forced-crack Propagation by Weld Nugget due to Absence of Continuous Grain Boundaries

41922 128X

Fig. 11. Cracks Radiating from the Weld Kernel

A thin zone of the cladding, approximately 0.001 in. thick, was found to be carburized in the bonded zone at the weld nuggets. No barrier was used to prevent the migration of carbon during the cladding process.

(2) Cladding from 2-in.-thick Practice Plate

The through cracking of the cladding on the practice plate followed a different mechanism, inasmuch as all cracks apparently started at the exposed surfaces.

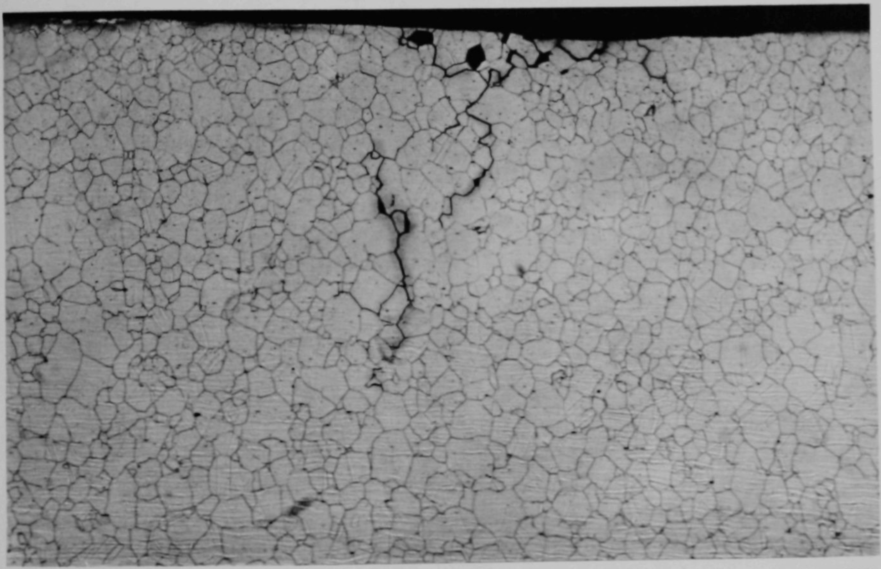
Radial cracking patterns around the kernels of the nuggets in the practice plate were similar to those seen in the vessel. Migration of carbon between the carbon steel and the stainless steel was less pronounced in the test plate, probably because the time at the 1150°F stress-relief temperature was considerably less than that for the reactor vessel.

(3) Cladding from 4-in. Lower Head Plate

This plate was a part of the heat of steel used in the construction of the vessel shell but was not subjected to the fabrication processes and the subsequent stress history of the reactor vessel. It was essentially "as-clad" SA-212-B material, which had been exposed to atmospheric environment during 4 to 5 years of unsheltered outdoor storage. Dye-penetrant examination of the cladding, as previously described, revealed that the surface was cracked to approximately the extent found in the cladding of the reactor vessel (steam zone).

Microscopic examinations of sections parallel and transverse to the nugget-rolling direction showed three types of internugget cracks. These were:

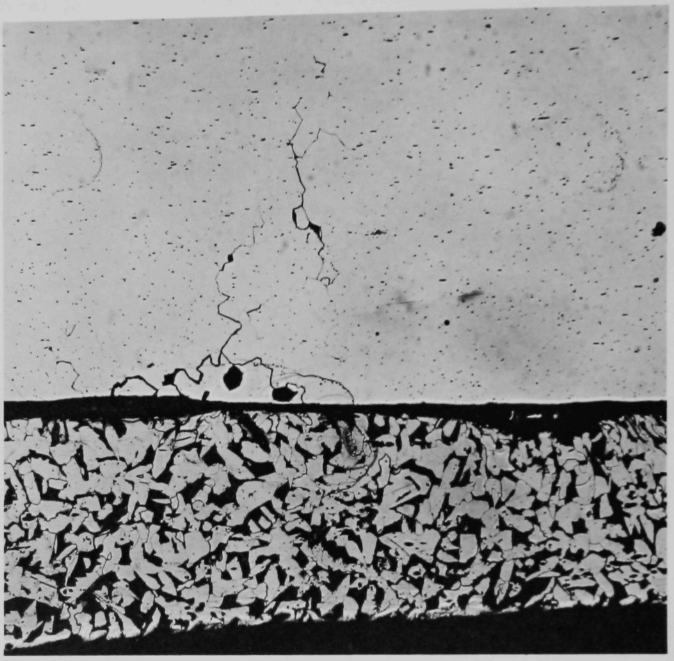
(a) Cracks originating at the surface and partially propagated toward the interface (Fig. 12).



41929 103X

Fig. 12. Cracks Originating at the Type 304 Stainless-steel Surface

(b) Cracks originating at the interface and partially propagated toward the surface (Fig. 13).



41934 103X

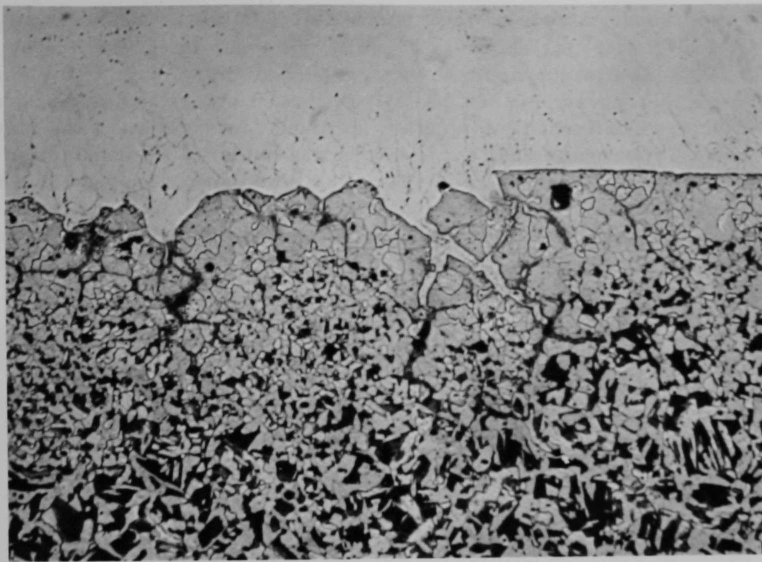
Fig. 13. Cracks Originating at the Interface and Partially Propagated toward the Surface

(c) Through cracks originating at the surface but none originating at the interface.

Except for variations in dimensions, structures in the nugget region were similar to the structures in the reactor vessel.

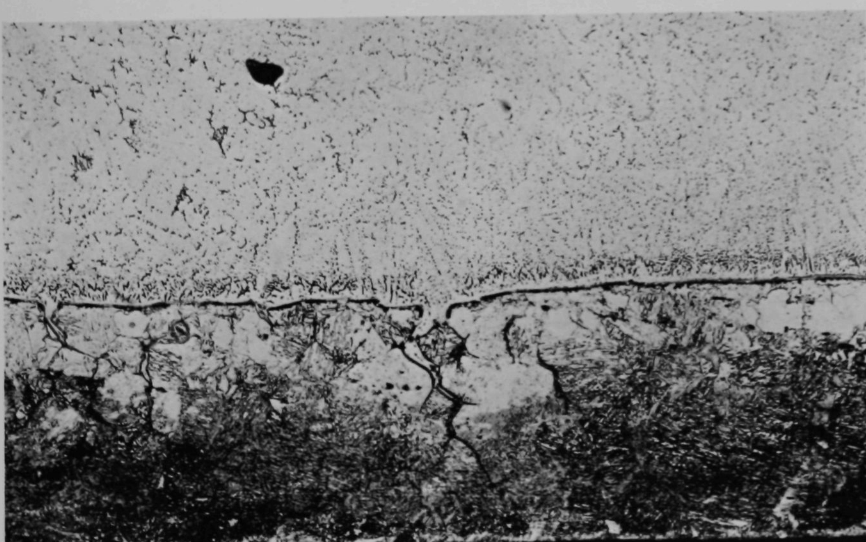
(4) SA-212-B Vessel Wall

Although the cracking of the vessel cladding was of great interest, all samples examined were searched primarily for evidence of concurrent cracking of the SA-212-B vessel plate. Several microcracks in the SA-212-B, ranging up to 0.008 in. in depth, were found. A number of the deeper microcracks were filled with intruded Type 304 stainless steel from the cladding process, as shown in Figs. 14 and 15. Many intrusions of molten stainless steel into oxide interfaces



42078 61X

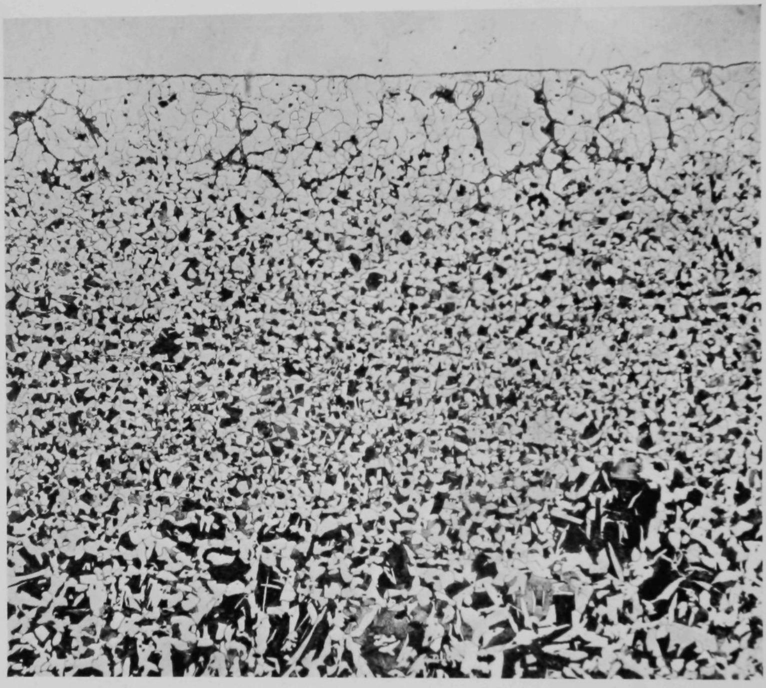
Fig. 14. Microcracks in Surface of SA-212-B Plate (Note intrusion of Type 304 stainless steel)



41935 103X

Fig. 15. Intrusion of Type 304 Stainless Steel into Pre-existing Microcracks of SA-212-B Plate

were also found. A zone of decarburization, 0.010 in. max, was found in the SA-212-B steel, as shown in Fig. 16. This was later verified by microhardness examination. Metallographic studies conclusively showed that these microcracks in the SA-212-B existed before the cladding was applied and were probably formed in the initial fabrication of the plate. None of the specimens examined showed any evidence of propagation of the surface microcracks as a result of the cladding process, vessel fabrication, or reactor operation. Propagation of these microcracks is not expected in the future operations of the vessel, which will be operated at the same or lower pressure and temperature levels as in the past.



42079 61X

Fig. 16. Decarburized Zone of SA-212-B Plate at Type 304 Stainless-steel Interface

b. General Observations

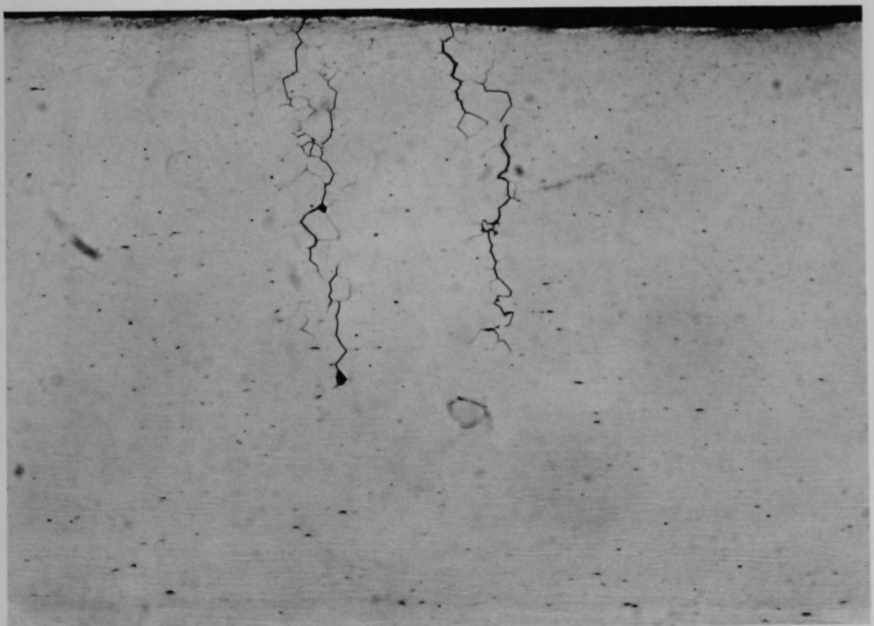
(1) The cladding cracks were intergranular, as shown in Figs. 17 and 18.

(2) None of the inter-nugget cladding cracks propagated across the unbonded boundary between the Type 304 stainless-steel cladding and the SA-212-B base plate. In the bonded region, all cracks in the wrought Type 304 stainless-steel cladding were stopped by the fusion boundary of the kernel.

(3) The actual bonded area between the clad and the base metal is less than 20% of the total interface area.

(4) The microcracking of the kernel is contained wholly inside the dendritic structure of the kernel.

(5) The wrought Type 304 stainless-steel cladding is sensitized. This occurred during the fabrication of the vessel. The equiaxed grains of the very fine-grained steel are prominently outlined by continuous networks of carbides, as shown in Fig. 19.

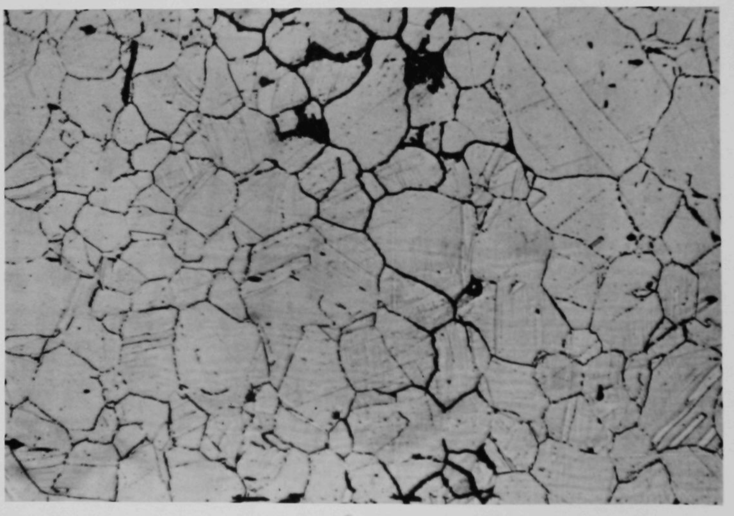


41923 128X

Fig. 17. Intergranular Cracking in Type 304 Stainless-steel Cladding

Fig. 18

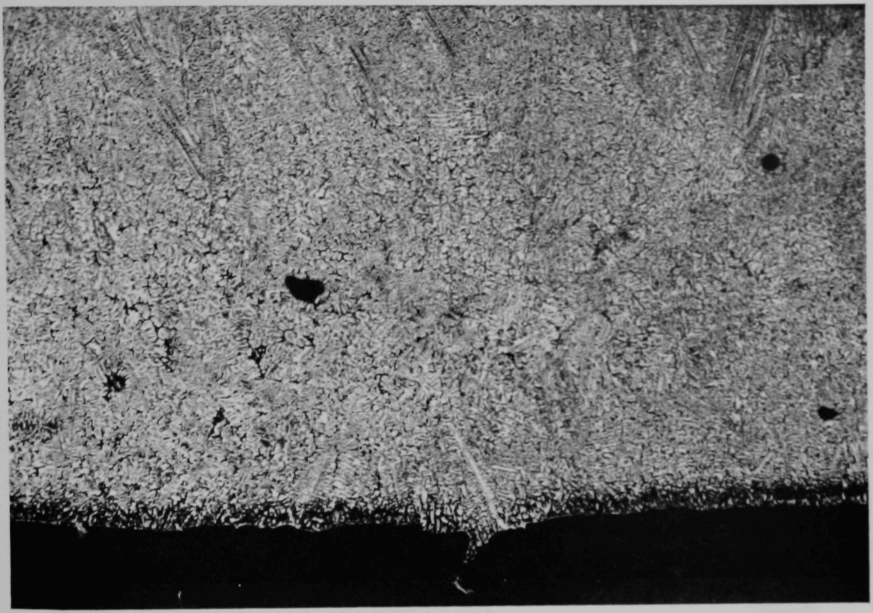
Intergranular Cracking in Type 304 Stainless-steel Cladding



250X

Fig. 19. Continuous Carbide Network in Type 304 Stainless Steel

(6) The fine dendritic structure of the kernel contains typical hot tears as well as rejected carbides, as shown in Fig. 20.



41928 103X

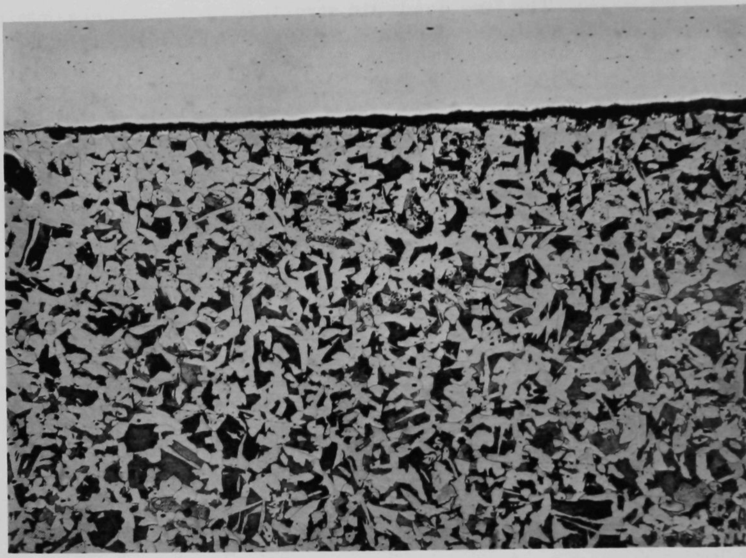
Fig. 20. Micro Hot Tears, Wholly Contained in Nugget

(7) The stainless-steel nugget (surface thumb print) is biphased and consists of an upper deformed wrought structure and a lower kernel cast structure.

(8) The general structure of the SA-212-B plate is that of a typical normalized and tempered carbon steel, as shown in Figs. 9, 13, and 21.

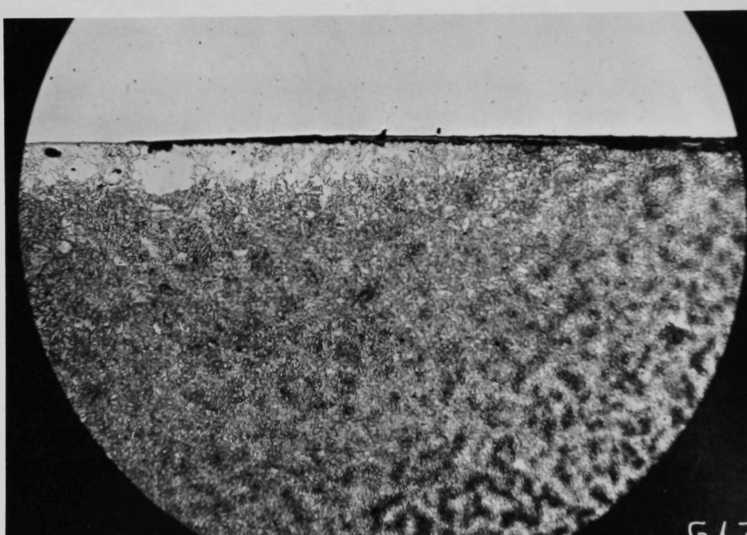
(9) In the bond region, the SA-212-B vessel steel was heated by the welding current (without melting) to the austenitizing temperature, quenched in situ by conduction into the interior of the carbon steel plate, and subsequently tempered. The transformed structure in the SA-212-B bonded area is shown in Figs. 14, 16, and 22.

(10) In the bond region, the SA-212-B vessel steel was decarburized to a depth of 0.010 in. max (see Figs. 15 and 16), and the SS Type 304 was carburized to a depth of approximately 0.001 in. (see Figs. 15 and 20).



42082 61X

Fig. 21. Typical Micro Structure of SA-212-B Plate from Reactor Vessel



42279 69X

Fig. 22. Typical Quenched and Tempered Structure of SA-212-B Plate at Bonded Area. Note that decarburization occurred after welding.

(11) No barrier was used in the cladding process to prevent the migration of carbon from the SA-212-B to the SS Type 304 cladding.

(12) The physical dimensions of the nugget on the exposed surface were fairly constant, but the kernel size varied.

4. Hardness Surveys

Microhardness surveys on the stainless-steel cladding and the SA-212-B steel of the pressure-vessel-boat samples were locally affected by the migration of carbon across the bond zone. The hardening of the stainless-steel cladding or the softening of the SA-212-B steel is not significant as far as the propagation of cracks or the strength of the vessel is concerned. Results of the hardness survey of both metals are summarized in Figs. 23, 24, and 25.

5. Induced Crack Propagation

Samples of the 2-in. practice plate, containing superficial surface cracks, were bent to force their propagation across the thickness of cladding. Interrupted slow bend (tensile) tests were used to propagate the embryo cracks under stress. For the two crack locations of interest, i.e., the inter-nugget crack and the nugget in-line crack, the SA-212-B steel was permanently bent to develop the crack in the clad material. Samples were approximately 3/4 in. thick (cladding plus SA-212-B steel).

Incremental bending of the sample containing an inter-nugget crack, propagated to the unbonded interface incrementally. Continued bending only opened the crack along its entire length without propagation across the interface.

Similar results were obtained from the forced development of an embryo crack in the wrought portion of the cladding overlaying the cast structure of the kernel. The crack was stopped by the fusion line of the kernel and is illustrated in Fig. 10.

D. Possible Cracking Mechanisms

Chemical, macroscopic, and microscopic examinations of the Type 304 stainless-steel cladding welded to the SA-212-B steel plate yielded much graphic and qualitative data.

Simple and straightforward calculations indicate that the most probable causes of cracking can be attributed, initially, to the differences in the behaviors of materials during the fabrication stage, and, later to low-cycle fatigue failures. Other failure mechanisms undoubtedly aided

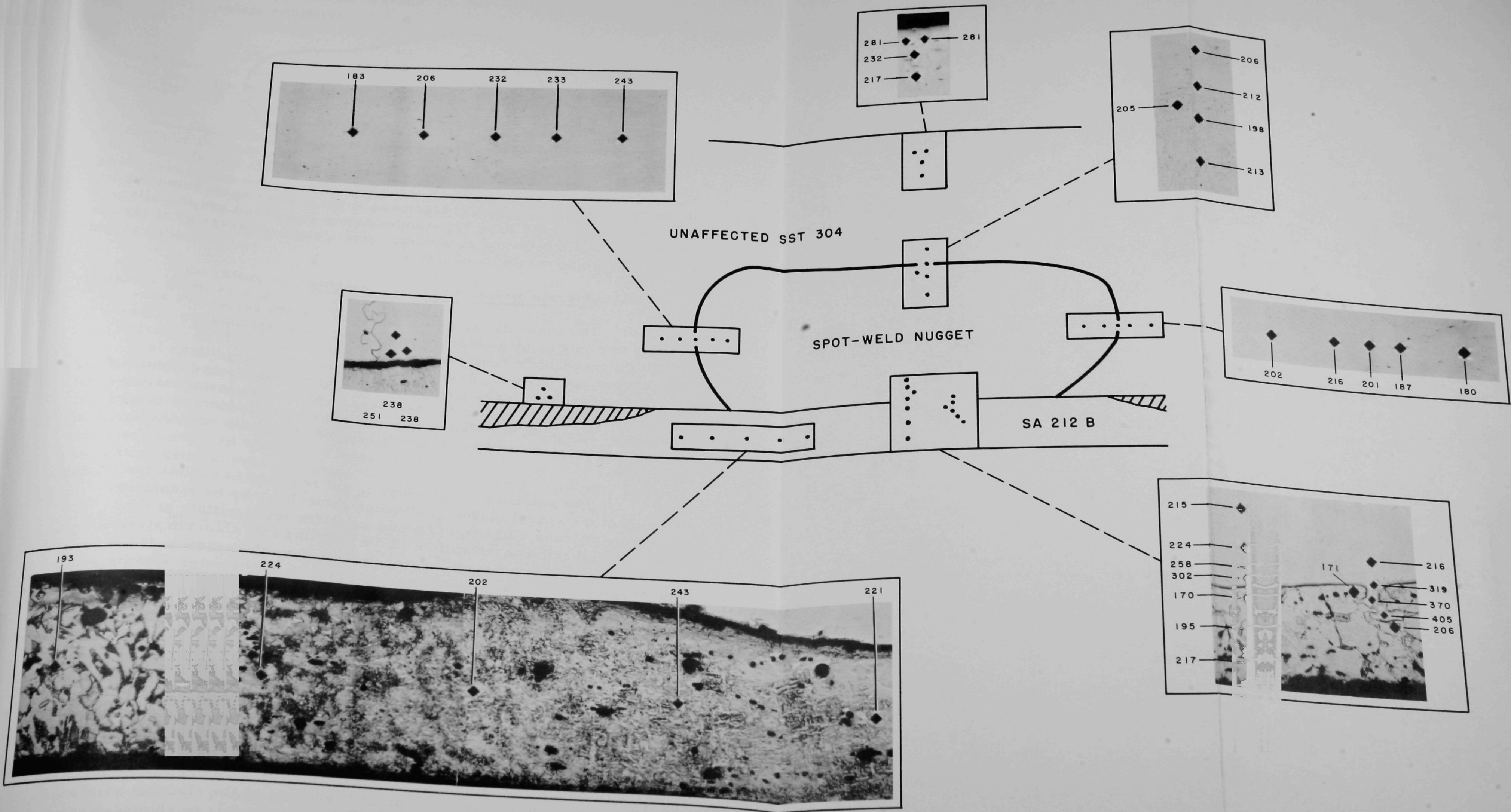


Fig. 24. Hardness Survey (DPH) of Boat Sample from Reactor Vessel. Note effect of carburization and decarburization, particularly at intrusions.

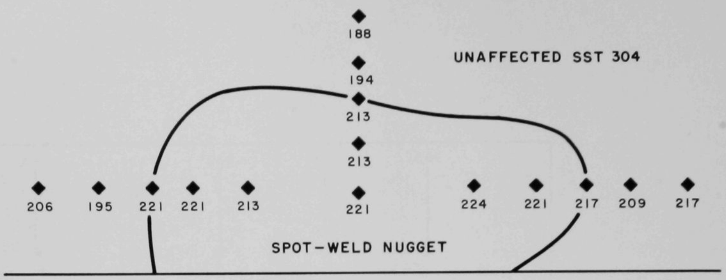


Fig. 25. Hardness Survey of Weld Nugget Area (DPH)

in the development of localized fractures of the reactor vessel cladding, inasmuch as all of the Type 304 stainless steel was sensitized initially by thermal heat treatments at the fabricator's plant. Discussion of the possible cracking mechanisms follows.

1. Chloride Stress-corrosion Cracking

The austenitic stainless steels of the 300 series are susceptible to cracking by a chloride stress-corrosion mechanism. Generally, the mechanism requires the simultaneous presence of chloride ions, oxygen, and stress. When cracking under these conditions is observed under the microscope, the classic branched "forked lightning" transgranular pattern is found. However, in some rare instances, intergranular cracking has also been reported; the mechanism of cracking in the unusual cases was obscured by abnormal grain boundary conditions.

The presence of chlorides is not required to explain the cracking of the vessel cladding, although chlorides may have contributed to the process. Stress-corrosion cracking must initiate at the locus of a chloride concentration. Even granting that chloride stress-corrosion cracking can occur as an intergranular phenomenon, it has been shown that the 2-in. practice plate cracked on the outer surface and propagated towards the interface. It has further been shown that much of the cracking of the actual reactor vessel cladding initiates at points other than at the interface.

Although high cladding stresses, oxygen, and abnormal grain boundaries existed simultaneously in the reactor vessel, the use of an ion-exchange purification system maintained reactor water that was essentially free of chloride ions. Thus, the exposed surface of the cladding was free of chloride ions. Chlorides may be present at the interface. Oxygen and water, however, are required to form the active, chloride stress-corrosion, cracking mechanism. This mechanism, to be effective at the interface, thus requires a prior breaching of the cladding barrier. Hence, despite the probable presence of small amounts of chlorides at the cladding interface, it is not believed that the chloride stress-corrosion cracking mechanism was a primary cause of the EBWR vessel cladding cracking because:

a. The cracks in the cladding of the 2-in. practice plate (which should be the most susceptible to stress-corrosion cracking by virtue of a moderately high chloride ion content at the carbon-steel/stainless-steel interface) originated at the surface.

b. One criterion for stress-corrosion cracking, when small amounts of chloride ions are present, is a means of concentrating the chlorides, as at a water/steam zone where the water carries chlorides which precipitate out on evaporation. This condition does not exist at the SA-212-B/Type 304 interface in the reactor vessel unless the cladding has already been breached, and even then the reactor primary coolant is so carefully controlled that the presence of chloride ions is highly unlikely.

c. Many cracks are internal with respect to the Type 304 stainless steel, extending neither to the surface or the interface.

2. Thermal-stress and Fatigue-failure Mechanisms

The most probable vessel-cladding cracking mechanisms are believed to be thermally induced mechanical-stress ruptures and low cycle fatigue failures. The ensuing analyses explain the various cracking modes (through-cracking, radial, and lamellar) and are based on metallographic evidence. The magnitudes of the multiaxial stress states are computed from the original internal memoranda on vessel fabrication details, heat-treatment furnace charts, and mill-test-report data.

a. Through-cracking of SS 304 Cladding

Through-cracking of the cladding in the vessel with propagation from the exposed surface to the carbon-steel interface, converse propagation, and simultaneous propagations from both surfaces were observed. Although there is no direct evidence for the mechanism, the ruptures are believed to be multiaxial thermal-stress ruptures generated in the spot-welding process and subsequent forming and heat-treating operations.

The model for the cladding-cracking mechanism comes from a consideration of the spot-welding process. In the process, linear rows of "thumb-print" nuggets were formed two at a time (about 16 in. apart) sequentially at short time and distance intervals. Adjacent rows were formed on the return portion of the welding machine cycle. In consequence, a thermal gradient was maintained between in-line nugget neighbors and neighboring rows throughout the cladding operation. The photomicrographic evidence for the rapid cooling of the melted cladding nugget is the fine-grained transformed structure of the stainless-steel kernel and the underlying transformed but unmelted structure of the carbon-steel plate at the bond interface. Since the thermal conductivity of the carbon-steel plate is greater than the overall heat-transfer coefficient to coolant water, the bond region was cooler and stronger than the heated and unmelted cladding above the kernel. Further cooling of the nugget increased tensile stresses until rupture occurred. Minor through-cracking of the flat stainless-steel panels subsequent to the cladding operation was detected and repaired by welding before the plates were formed into the vessel configuration.

b. Radial Cracking from Weld Nugget

Longitudinal and transverse-section micrographs show that the kernel of the nugget is oblate. This kernel was formed when the stainless steel was melted locally by the passage of welding current. Its unbonded boundaries are substantially larger than those of the bond area. The kernel is surrounded by unmelted wrought SS 304 and the carbon-steel base plate. In essence, this constituted a contained hydraulic system whose boundaries were ruptured by a molten stainless-steel core. The failure of a membrane by hydrostatic pressure is radial, with cracks substantially normal to the surface, as observed in Fig. 26. Also observed were extrusions of molten cladding material into the microvoids at the SA-212-B boundary, which provided a partial relief of the pressure forces.

41919 128X

Fig. 26. Radial Cracks at Interface between Weld Nugget and Unaltered Type 304 Stainless Steel

c. Lamellar Cracking of SS 304 Cladding

Lamellar cracking of the cladding was observed in the vessel shell boat samples and in the 2-in.-thick practice plate. Although there is no direct metallographic evidence for shear-type failures, the ruptures are believed to be associated with the "normalizing" heat treatment of the SA-212-B plate. The heat treatment consists of heating the cladded carbon-steel plate to the austenitizing temperature of the SA-212-B steel (1600/1650°F), followed by cooling in still air. The heat treatment yields a steel with a uniform structure.

At the normalizing temperature, both the carbon steel and stainless steel are plastic, and since holding time at temperature is long (one hour per inch of thickness), residual stresses are relaxed. In the cooling process, both steels contract, but not uniformly; the coefficient of expansion is greater for the SS 304 than for the carbon steel. Tensile stresses are induced in SS 304, while the base carbon steel is stressed in compression; stress intensities are inversely proportional to the thicknesses of the two members joined.

Since the members are joined by the spot welds, the tensile cladding forces are transmitted to the much thicker backing plate by bending moments. The stiffness of the beam sections are proportional to the square of the thickness.

A simplified analysis of a composite beam consisting of a 0.109-in.-thick SS 304 cladding attached to the $2\frac{7}{16}$-in. carbon-steel backing, shows that the welded-on (resistance-spot) cladding will rupture in tension when cooled from the normalizing temperature. The computed stresses are as follows:

$$\sigma_t = \frac{E_1\alpha_1\Delta t}{2(1-\mu)} - \frac{E_2\alpha_2\Delta t}{2(1-\mu)},$$

where

σ_t = thermal stress

E_1 and E_2 = modulus of elasticity of SA-212-B carbon steel and SS 304 cladding, respectively,

= 28×10^6 psi;

α_1 = coefficient of expansion of carbon steel

= 6.3×10^{-6} in./in./°F (at room temperature);

$$\alpha_2 = \text{coefficient of expansion of SS 304 cladding}$$

$$= 9.5 \times 10^{-6} \text{ in./in./°F (at room temperature);}$$

$$\mu = \text{Poisson's ratio}$$

$$= 0.3 \text{ (for both SA-212-B and SS 304 steels);}$$

and

$$\Delta t = 1540°\text{F (cooling from 1600 to 60°F).}$$

The thermal-stress equation reduces to

$$\sigma_t = -0.715 \, E \Delta\alpha \Delta t,$$

or

$$\sigma_t = -98{,}500 \text{ psi (tension).}$$

The computed value of tensile stress in the SS 304 cladding is above the short-time ultimate tensile strength of the steel over the temperature range of interest.

The compressive stress in the base carbon-steel plate is negligible, approximately one-eighth of the yield strength, viz.,

$$\sigma_c = 98{,}500 \times \frac{0.109}{2.437} = 4410 \text{ psi (compression).}$$

The ratio of the stiffnesses of the beam elements, in bending is

$$\left(\frac{2.437}{0.109}\right)^2 = 498.$$

This ratio is far in excess of the ratio of changes in modulus of elasticity over the temperature range considered, and the vessel shell plate is a "strong back" which loads the SS 304 cladding under all cooling conditions below 1600°F.

Since bending moments in a beam are transmitted by horizontal shear forces, the observed lamellar cracks in the SS 304 cladding are shear failure manifestations of beams loaded to destruction.

The metallographic findings and calculations thus support our conclusions of the nature of cladding cracking, i.e., stress rupture. The calculations also explain the reasons for the necessity of repairs to the cladding in the fabricator's shops before shipment of the vessel. The metallographic examination and calculations therefore emphasized the need for establishing the crack-arrest properties of the cladding-to-shell boundary and which were explored in laboratory bend tests previously described.

The following calculation reveals that the reactor start-up and shutdown cycles superimposed a low-cycle fatigue mechanism over the high stress state of the cladding, which explains the presence of cracks (post-100-MWt operation) and their apparent absence in the 1959 modification of the EBWR. When the reactor is brought up to its operating pressure and temperature of 600 psig and 489°F, respectively, the reactor shell and cladding are:

(1) Stressed elastically (at pressure), and

(2) Stressed thermally.

The elastic pressure stress is, approximately, 12,500 psi (tension).

The differential thermal stress in the cladding is:

$$\sigma = \frac{E\Delta\alpha\Delta t}{2(1-\mu)}$$

$$= \frac{28 \times 10^6 \times 3.5 \times 10^{-6}(489 - 60)}{2(1 - 0.3)}$$

$$= 30{,}000 \text{ psi (compression)}.$$

The net stress range is +30,000 - 12,500 = +17,500 psi, since the cladding expansion is greater than the elastic expansion of the steel to which the cladding is attached by the process of resistance spot welding. As the reactor heats up, tensile stresses are reduced from the ultimate (at room temperature) by 17,500 psi, and, to a value below the ultimate strength at the hot (489°F) condition. Thus, cracks are propagated by cyclic fatigue, which is time-dependent. From this calculation, it is predicted that additional cladding cracks will appear on the surface in the future, propagated from unsurfaced cracks, which are metallographically known to be present.

It is believed that the mechanism exerting the greatest influence on the observed cracking is the latter. Contributions from the chloride-ion stress-corrosion cracking mechanism following the breaching of the cladding are improbable.

III. EVALUATION

A. Fracture Analysis

It has been demonstrated that all crack propagation in the stainless-steel cladding is intergranular and that, in fact, propagation is arrested where chromium carbide grain boundaries do not exist (see Fig. 10). No propagation of cracks across the stainless- and carbon-steel interface has been observed nor is to be expected since there is no continuity of the stainless-steel grain boundaries in the weld nugget nor in the unbonded areas.

A thorough review has been made of the potential effect of defects in the SA-212-B plate on reactor vessel integrity, including gross rupture type failure. The surface of the SA-212-B plate on the interface side is, in all observed areas, free from significant defects. Some decarburizing to a depth of 50-250 microns is evident in the bonded region. This decarburization is unimportant in that it has no bearing on either the strength of the $2\frac{7}{16}$-in.-thick reactor vessel plate nor on the fracture-mechanics hypothesis related thereto.

It has been shown that the NDT[5a] (nil ductility temperature) of the plate used for this vessel is about 80°F. On this basis, the FTP (fracture transition, plastic) of this material is 200°F and the FTE (fracture transition, elastic) is 140°F. Figure 27 indicates[6] that, at NDT + 60°F, a crack 2 ft long stressed at the yield strength will not propagate.

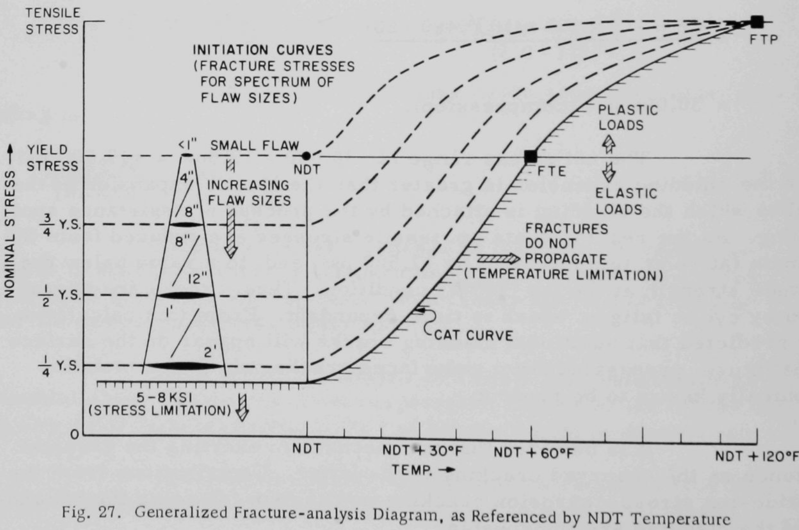


Fig. 27. Generalized Fracture-analysis Diagram, as Referenced by NDT Temperature

The only mechanism of crack growth above CAT (crack arrest temperature) at stresses less than yield is fatigue. Since neither mechanical nor thermal stress cycling is of sufficient magnitude or frequency to achieve this growth at temperatures of operation (which are above CAT), it is not reasonable to assume crack propagation in the vessel plate. The ASME Boiler and Pressure Vessel Code, Section III, Appendix 1, gives the minimum thickness requirement as

$$t = \frac{PR}{S_m - 0.5P}.$$

Under EBWR conditions,

$$S_m = \frac{P(R + 0.5t)}{t} \cong 11{,}500 \text{ psi, or}$$

25.3% of Y.S. based on actual mill tests

or

30.3% of Y.S. based on minimum specifications,

which represents a

factor of safety of $6\frac{2}{3}$ based on actual T.S. as determined by mill tests

or a

factor of safety of $5\frac{1}{4}$ based on minimum T.S. from specifications.

Also considered was the effect of irradiation on NDT. It is estimated that the maximum nvt >1 MeV to which any part of the EBWR vessel has been exposed is 8.65×10^{17}, and the projected nvt >1 MeV for 80 MW-yr (the maximum anticipated during the Plutonium Recycle Program) is $<3 \times 10^{18}$. It has been shown, and it is accepted generally, that there is an insignificant effect of irradiation on NDT of SA-212-B steel resulting from exposure of nvt >1 MeV up to 10^{18}. Figures 28 and 29 indicate[7] that, at nvt >1 MeV = 3×10^{18}, the NDT increases about 150°F, which would result in FTE of 290 or 200°F below operating temperature of EBWR. This subject is discussed in more detail in Appendix B of the "Safety Analysis Associated with the Plutonium Recycle Experiment in EBWR."[8]

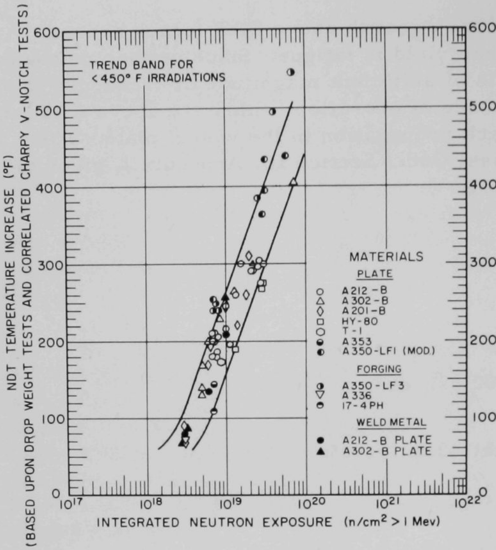


Fig. 28

Increase in NDT Temperatures of Steels Resulting from Irradiation at Temperatures below $450^{o}F$

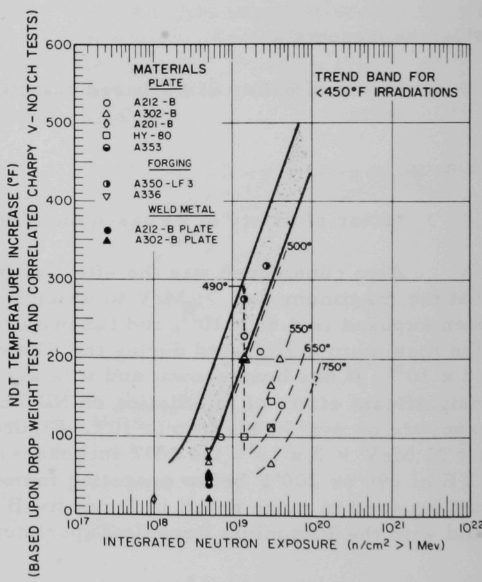


Fig. 29

Increase in NDT Temperatures of Steels Resulting from Irradiation at Temperatures above $450^{o}F$. Points at 5×10^{18} n/ cm^2 represent early irradiations in the Brookhaven Graphite Reactor at 500 to $600^{o}F$.

B. Corrosion of SA-212-B

Inasmuch as it is proposed to operate the pressure vessel without recladding the areas of SA-212-B laid bare as a result of grinding away through cracks in the cladding and taking boat and strip samples, the possible corrosion effects have been evaluated.

1. The general corrosion of SA-212-B in 500°F oxygenated water[9] is about 400 $mg/dm^2/mo$, or about 0.0025 in./yr (2.5 mpy). Since the vessel walls are 9/16 in. (in the gouged areas) thicker than dictated by the ASME Boiler and Pressure Vessel Code for 650-psi design, many times the anticipated general corrosion rate can be accommodated.

2. While the corrosion of carbon steel coupled to austenitic stainless steel might be expected to accelerate due to galvanic effects, particularly in view of the relatively small area of SA-212-B exposed, it must be kept in mind that the working fluid is a poor conductor. In April 1958, Berry and Fink of BMI, in a report to the International Nickel Company, Inc., stated, "Galvanic coupling of Inconel to steel (carbon) or to Type 347 stainless steel does not accelerate attack on either member of the couple when exposed to reactor waters." Since the potential (negative to saturated calomel) of Inconel is essentially the same as that of the 300 series stainless steels, it would appear that no deleterious galvanic corrosion effects will occur between the stainless steel and the SA-212-B plate.

3. Some pitting corrosion is possible in the crevices formed by the interface of the SA-212-B steel with the Type 304 stainless steel. The role of dissolved oxygen is believed to be the controlling factor in pitting. If oxygen is present in small amounts, and the corroding solution is stagnant, passivity may not be preserved over the entire surface and pits may develop. If the concentration of oxygen is sufficient to maintain an unbroken passive film over the entire surface, pits will not form. In the absence of oxygen or some other oxidizing agent, the cathodic areas will not be depolarized and the development of pits will be arrested.

It is therefore concluded that pitting, if any, will be self-limiting (due to broadening into "dimples" or by ultimate exclusion of all corrodent from the bottom of the pit) at shallow depths. In addition, the thickness of the SA-212-B steel wall of the reactor pressure vessel exceeds Code requirements by about 9/16 in. Thus, prolonged operation can be sustained at the general corrosion rate of 0.0025 in./yr.

C. Change in Design Pressure Rating

The integrity of the EBWR pressure vessel is augmented by a change in the design pressure rating, brought about by circumstances in no way related to the cracking of the vessel cladding or, for that matter, to the vessel in any way.

During the examination of the EBWR primary piping system, it was established that the 8-in. reboiler piping was SA-358 welded pipe in its entirety. The maximum pressure limit was calculated by using the equations of the 1955 ASA B.31.1 Power Piping Code. The maximum allowable pressure was found to be 750 psig, when a C constant, 0.065 in., was included in the wall-thickness calculation. Although this computed pipe wall was just under the thickness required to meet ASA Piping Code interpretation for 800 psig, the pipe was accepted in 1960 for the following reasons:

1. Welded pipe has a more uniform wall thickness and concentricity than a seamless pipe.

2. Welded pipe was immediately available.

3. The pipe installed was spot X-rayed.

4. Mill hydrostatic test at 1500 psig was performed on the pipe.

Because the pipe was radiographed and its strength was demonstrated by the abnormally high mill-hydrotest pressure, the requirements of Section VIII of the ASME Code were met for 800-psi service.

System design pressure was, however, reduced from 800 to 650 psig in view of the limitations of the two Codes and because neither the prior operating history of EBWR nor the plans for the future operating of the reactor plant exceed 600 psig. The rerating was accomplished by a substitution of valve springs to two of the safety valves already in the system. This decision was made despite the fact that the entire primary system, which included the reboiler system, was tested satisfactorily in the past at 1200 psig.

IV. REACTOR VESSEL SURVEILLANCE PROGRAM

The EBWR reactor vessel contains cladding cracks and intentional cladding defects resulting from the investigative procedures. This has resulted in exposure of the SA-212-B carbon-steel vessel to the reactor coolant. It is reasonable to assume that the previous operation of the reactor included operation with cracked cladding in the reactor vessel. Subsequent inspection has revealed no deleterious effect of such operation and no evidence of crack propagation into the carbon-steel vessel wall and/or unacceptable corrosion of the vessel wall. During future operation of the reactor, a surveillance program will be employed on the reactor vessel to insure that this situation continues.

This surveillance program will recognize that the condition of the reactor vessel may have been changed as a result of the exposure of the vessel where it has been laid bare by the removal of cladding. It will also reflect that only the upper half of the vessel is accessible for inspection.

A. Preparation of Vessel for Surveillance

To detect structural changes in the cladding and the carbon-steel vessel wall during future operations, a reference condition has to be established. Surveillance will be conducted to determine what is happening to the cladding and its significance on continued reactor operation; to determine whether any cracking develops in the SA-212-B carbon steel vessel wall; and to determine the amount and significance of corrosion.

Dye-penetrant testing will be utilized to identify new cracking by comparison with photographic records of previous conditions.

An optical depth micrometer has been used to establish reference points for the SA-212-B steel prior to reactor operation. During scheduled shutdown periods, corrosion will be measured and compared with the reference measurements.

Inspections will be conducted in four specific areas of the upper course of the vessel; in two panels (one in the water zone, one in the steam zone) that show essentially no indication of surface cracks, and in two panels (in these zones) that do show evidence of surface cracks. To facilitate this surveillance, the lower section of the upper shock shield (which covers the clad region in the water zone) has been modified by installation of two removable panels over the areas selected for surveillance. Sufficient area of the steam zone can be inspected without removing any vessel components.

B. Cladding Surveillance

The upper course of the reactor vessel has had cladding removed (by grinding) in various areas, as indicated in Fig. 1. All unclad areas have been photographically documented for reference and comparison in future inspections. Any unusual conditions that develop will be evaluated.

The incidence of cladding cracking in the steam zone (upper quarter of the vessel) is several times greater than that in the water zone. In the latter area, cracking incidence diminished as the belt line was approached. Only two incidents of cracking were found in the region about 6 in. above the circumferential weld located near the center of the vessel.

Only the upper half of the vessel is accessible for inspection; the lower half of the vessel is made inaccessible by a thermal shield and core components, as well as being highly radioactive. Any cladding cracking that may exist in the lower half of the vessel is believed to be no greater than that found in the water zone. Therefore, surveillance of the upper half of the vessel, including the steam zone as well as the water zone, is believed to provide adequate indication of the condition of the lower half.

C. SA-212-B Vessel Steel Surveillance

Exposed SA-212-B reactor vessel material in the surveillance areas will be retested with dye-penetrant to determine if cracking has been initiated. Any cracks uncovered will be cause for re-evaluation.

The general family of carbon steels, which includes the SA-212-B vessel steel, corrode slowly in water-reactor environments. Because their corrosion products do not adhere to the corroded surface, they will be released to the coolant and circulated through the core. However, no gross corrosion of the SA-212-B steel behind the cracked cladding has been found. The safety of the vessel for the planned operating pressure (600 psig) will be preserved because extra thickness of SA-212-B metal wall is available for corrosion. The ASME Code (Section I) requirement for minimum thickness of the vessel for the 600-psi operations (with safety valves set to open at 650 psi) is given by

$$t_m = \frac{PD}{2SE - 0.6P} + 0.1,$$

where

P = 650 psig,

S = 17,500 psi,

E = 1.0 (for plate),

and

$$D = 84.625 \text{ in. (max diam at root of boat sample gouge),}$$

so that

$$t_m = \frac{650 \times 84.625}{2 \times 17{,}500 \times 1.0 - 0.6 \times 650} + 0.1$$

$$= 1.69 \text{ in.}$$

The minimum wall thickness, at the root of the boat sample gouge, is actual plate thickness minus the depth of gouge, or

$$t_{actual} = 2\tfrac{7}{16} - 3/16 = 2\tfrac{1}{4} \text{ in.}$$

The excess metal in the wall is 0.56 in. (~9/16 in.). Although the explored cracked length is large, the total exposed SA-212-B area is small: 40 ft x 1/4 in. wide = approximately 1 ft^2. Another equivalent amount was uncovered when strip and boat samples were removed from the wall. Thus the total exposed area is less than 1/2% of the internal vessel area.

D. Corrosion Surveillance

1. General Corrosion

The general corrosion of SA-212-B in 500°F oxygenated water[9] is about 400 $mg\text{-}dm^{-2}\text{-}mo^{-1}$, or about 0.0025 in./yr. It is proposed that, if a rate of 1000 $mg\text{-}dm^{-2}\text{-}mo^{-1}$ (arbitrarily selected) is exceeded, the effects of corrosion will be re-evaluated prior to continued operation. Since the EBWR vessel walls are 9/16 in. thicker than required by the ASME Boiler and Pressure Vessel Code, prolonged operation with these corrosion rates can be tolerated.

Reference points have been established in the vessel wall to permit measurement of the corrosion rates by utilizing an optical micrometer sensitive to 0.001 in. Differences in the profiles before and after periods of operation will yield the corrosion rates in inches per year.

2. Pitting Attack

Pitting of SS Type 304 cladding has not been a problem in the past. Although pitting of the stainless cladding is not expected, accessible sections will be scrutinized periodically during the program for any indication of pitting. If pitting occurs, measurements will be made to establish rates.

In future operations, areas of the SA-212-B carbon-steel vessel will be exposed to the reactor coolant wherever cladding was ground away to remove the cracks. Periodic inspection of the cladding during the grinding operation was carried on at the same time to determine the nature of the cracks. The channels ground out during this operation are about 1/4 in. wide and account for about 40 lineal feet. Thus, a similar amount of carbon steel will be exposed to the reactor water. Such surface discontinuities occur in both the steam and the water zones of the upper half of the vessel.

The exposed SA-212-B steel might be attacked by pitting corrosion. This type of corrosion is influenced by temperature, oxygen concentration, and the presence of water. The role of dissolved oxygen is believed to be the controlling factor in pitting. If oxygen is present in small amounts, and the water or steam is stagnant, passivity may not be preserved over the entire surface and pits might develop. If the oxygen concentration is sufficient to maintain an unbroken, passive film over the entire surface, pits will not form. The consensus of corrosion experts discounts serious damage by pitting attack; pitting will be self-defeating by broadening into "dimples," or by the exclusion of the corrodent from the bottom of the pit.

Direct measurements will be made to determine the depth and contours of any pits. At each inspection period, loose corrosion products (Fe_2O_3) will be brushed out prior to obtaining measurements with an optical depth micrometer. A 1/4-in. maximum reduction of the wall thickness by local pitting during the Plutonium Recycle Program is considered acceptable. However, if pitting exceeds 1/8 in. in depth between consecutive inspection periods, corrosive attack will be re-evaluated.

3. Galvanic Corrosion

Galvanic corrosion may be expected whenever two dissimilar welded metals form a closed electrical circuit. The severity of the attack is related to the voltage gradients developed in a particular system and the resistance of the electrolyte. The form of the attack may be either general (as in a lead storage battery) or local pitting.

In the EBWR, water purity is high except in special circumstances which require the introduction of small amounts of boric acid. Since high-purity water is a good insulator, galvanic corrosion does not appear to be a problem. However, if galvanic-corrosion effects are noted, the extent of the penetration into the interface area and into the SA-212-B material will be explored. The limiting penetration into the carbon steel shall not exceed that established for pitting corrosion.

E. Schedule

All the disturbed vessel surfaces were conditioned by grinding to eliminate sharp edges and notches in order to eliminate stress points and to establish an initial reference point for the vessel surveillance program. After the vessel is loaded with its core, subsequent examinations are scheduled at approximately 6-month intervals to establish corrosion rates and incidence of cladding cracking.

The inspections will include the unobstructed walls of the vessel in the steam zone above the shock shield, and localized areas in the water zone made accessible by a modification of the shock shield. Dye-penetrant examinations for bringing out details of any cladding cracks and vessel cracks will be the primary inspection tool. Corrosion rates will be measured with an optical micrometer.

Half couplings welded to the panels of the cladding for the high-pressure gas tests of cladding integrity have been retained for possible future use; the couplings were plugged to exclude any particulate matter.

F. Water-chemistry Program

The area of exposed SA-212-B steel within the reactor vessel is small ($<0.5\%$ of the vessel internal area). Since its chemical composition is different from that of the cladding, it is believed that any significant release of vessel corrosion products to the water can be detected.

Newly-installed high-capacity pumps in the ion-exchange system are expected to remove corrosion products from the water at a faster rate. This increased scavenging rate will reduce the previously established background and will aid the vessel-corrosion studies. Periodic monitoring of water purity, pH, gross activity, and crud analyses will be carried out throughout the Plutonium Recycle Program.

V. CONCLUSIONS

Cracks were found in the cladding of the EBWR reactor vessel. Visual and dye-penetrant inspections were made on numerous clad surfaces as well as exposed surfaces of the carbon steel laid bare by the removal of cladding. Boat samples removed from the reactor vessel were examined in detail. Bend tests were performed on boat samples. There was no indication of crack propagation across the interface in all the examinations and tests performed. These findings are further supported by calculations that show the SA-212-B carbon steel to be the "strong back" of the clad plate, and that the cladding will rupture in tension.

The most probable cause for the cracking in the cladding can be attributed, initially, to the differences in the behavior of the two metals during the fabrication stage, and later to low-cycle fatigue failures.

Regardless of the actual mode of fracture in the cladding, it has been shown, through nondestructive and destructive tests, that it is highly improbable that cracks in the stainless steel will propagate into the SA-212-B carbon-steel plate. Furthermore, cracking of the cladding appears to have originated with the cladding process and has existed during the entire operation of the reactor. The cladding never contributed a strength factor to the vessel, nor did its cracking create any identifiable deleterious effects.

The removal of cladding as a part of the vessel examination resulted in exposure of the interior of the SA-212-B carbon-steel vessel to the reactor environment. Although the corrosion of the carbon steel may increase in those areas, it is considered highly improbable that the increase can be of sufficient magnitude to impair the integrity of the vessel.

The lower half of the vessel and the lower head were not examined because of high radiation levels and inaccessibility imposed by in-place core support structure and the boron stainless-steel thermal shield. It was observed, however, that cracking of the cladding was most severe in the steam zone of the vessel, with progressively less cracking occurring below the water line and lower areas. It is assumed, a priori, that the cladding in the lower section of the pressure vessel is no worse than the condition of the cladding examined in the water zone.

It is concluded that the reactor-vessel integrity has not been impaired significantly, and that the reactor vessel is adequate for continued operation.

ACKNOWLEDGMENT

The authors gratefully acknowledge the assistance of O. J. Daniel for his supervision and guidance in obtaining boat and strip samples, and to S. Greenberg for his able assistance in identifying corrosion products during the vessel inspection program.

REFERENCES

1. The Experimental Boiling Water Reactor, ANL-5607 (May 1957).

2. Reactor Engineering Division Quarterly Report - Section I - October, November, December 1955, ANL-5561 (April 1956).

3. Reactor Engineering Division Quarterly Report - Section I - January, February, March 1956, ANL-5571 (July 1956).

4. Reactor Engineering Division Quarterly Report - Section II - April, May, June 1956, ANL-5601 (Dec 1956).

5. S. P. Rideout, Stress Corrosion Cracking of Type 304 Stainless Steel in High Purity Water, 2nd Int. Conf. on Metallic Corrosion, New York (March 1963).

5a. EBWR Test Reports (100-MWt Operation), ANL-6703 (Jan 1964), p. 531

6. W. S. Pellini and P. P. Puzak, Fracture Analysis Diagram Procedures for the Fracture-Safe Engineering Design of Steel Structures, NRL-5920 (March 15, 1963).

7. L. E. Steele and J. R. Hawthorne, New Information on Neutron Embrittlement and Embrittlement Relief of Reactor Pressure Vessel Steels, NRL-6160 (Aug. 7, 1964).

8. B. J. Toppel, P. J. Vogelberger, Jr., and E. A. Wimunc, Safety Analysis Associated with the Plutonium Recycle Experiment in EBWR, ANL-6841 (to be published).

9. Corrosion and Wear Handbook (for Water Cooled Reactors), TID-7006, p. 110 (March 1957).